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DUANE ARNOLD

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10 CFR 50.69

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Duane Arnold Energy Center
Docket No. 50-331
Renewed Facility License No. DPR-49

License Amendment Request (TSCR-195): Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"

Pursuant to 10 CFR 50.90, NextEra Energy Duane Arnold, LLC (NEDA) is submitting a license amendment request (LAR) to revise the Duane Arnold Energy Center (DAEC) Renewed Facility License (RFL). The proposed amendment would modify the DAEC licensing basis, by the addition of a license condition, to allow for the implementation of the provisions of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows an improved focus on equipment that has greater safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the DAEC RFL. The categorization process being implemented through this change is consistent with Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006. Attachment 1 of the enclosure provides a list of categorization prerequisites. Use of the categorization process on a plant system will only occur after these prerequisites are met.

Though routine maintenance updates have been applied, the PRA models supporting this application have previously been reviewed by the NRC and were found technically adequate for regulatory application. The technical adequacy of the internal events, including internal flooding, PRA model was previously evaluated by the NRC as part of approving adoption of the Surveillance Frequency Control Program. The technical adequacy of the fire PRA model was previously evaluated as part of the NRC's approval of DAEC's transition to National Fire Protection Association (NFPA) 805. NEDA requests that the NRC utilize the results of the reviews of the PRA technical adequacy for those applications when performing the review for this application.

This letter contains no new and no revised regulatory commitments.

In accordance with 10 CFR 50.91, a copy of this application with the enclosure is being provided to the designated State of Iowa official.

As discussed in the enclosure, the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change. The DAEC Safety Review Group has reviewed the proposed license amendment.

If you have any questions or require additional information, please contact Robert Murrell, Licensing Project Manager, at 319-851-9496.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 11, 2026

Kenneth Mack

Kenneth A. Mack
Director, Licensing and Regulatory Compliance

Enclosure: Evaluation of the Proposed Change
Attachments (6)

cc: NRC Region III Administrator
NRC Decommissioning Inspector
NRC NMSS Project Manager
Designated Iowa State Official

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

License Amendment Request (TSCR-195): Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"

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1. SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), requirements will not be changed or will be enhanced. This allows an improved focus on equipment that has greater safety significance resulting in improved plant safety.

2. DETAILED DESCRIPTION

2.1 Current Regulatory Requirements

The Nuclear Regulatory Commission (NRC) has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. The structures, systems and components (SSCs) necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related" and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 Reason for the Proposed Change

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, probabilistic risk assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an

extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available using PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be LSS, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be HSS, requirements will not be changed or will be enhanced. This allows an improved focus on equipment that has greater safety significance resulting in improved plant safety.

The rule contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference 1), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to assure functionality and reliability are maintained and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to adjust the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as HSS, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable confidence (which is a reduced level compared to the reasonable assurance criteria used for many special treatments) that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow NextEra Energy Duane Arnold, LLC to improve focus on equipment that has greater safety significance resulting in improved plant safety.

2.3 Description of the Proposed Change

NextEra Energy Duane Arnold, LLC (NEDA) proposes the addition of the following condition to the Renewed Facility License (RFL) of Duane Arnold Energy Center (DAEC) to document the NRC's approval of the use of 10 CFR 50.69.

> NextEra Energy Duane Arnold, LLC (NEDA) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown

risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2, Class 3, and non-class SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach described in NEDA's submittal letter dated [DATE]; as specified in License Amendment No. [XXX] dated [DATE].

NEDA is approved to apply the ANO-2 passive categorization methodology solely to the Control Building Chiller system for purposes of 10 CFR 50.69 categorization. NEDA is also approved to transition to the Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components, as described in NEDA's letter dated [DATE], and as approved in License Amendment No. [XXX] dated [DATE]. The enhanced methodology supports categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs by performing a plant-level evaluation of the pressure retention function across all systems, treating the pressure boundary function as a system for 10 CFR 50.69 categorization purposes. Until completion of the plant-level evaluation using the enhanced methodology, the categorization of the Control Building Chiller system performed using the ANO-2 passive categorization methodology will remain valid. Upon completion of the plant-level evaluation using the enhanced methodology, the ANO-2 approach will no longer be used for 10 CFR 50.69 categorization purposes.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

3. TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
>
> (i) A description of the process for categorization of RISC–1, RISC–2, RISC–3 and RISC–4 SSCs.
>
> (ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).
>
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and

externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

Each of these submittal requirements is addressed in the following sections.

Though routine maintenance updates have been applied, the PRA models supporting this application have previously been reviewed by the NRC and were found technically adequate for regulatory application. The technical adequacy of the internal events, including internal flooding, PRA model was previously evaluated as part of adopting the Surveillance Frequency Control Program (SFCP), which was approved by the NRC on February 24, 2012 (Reference 2). The fire PRA model was evaluated as part of transitioning to National Fire Protection Association (NFPA) 805, which was approved by the NRC on September 10, 2013 (Reference 3). NEDA requests that the NRC utilize the results of the reviews of the PRA technical adequacy for those applications when performing the review for this application.

3.1 Categorization Process Description (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

NEDA will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference 4). NEI 00-04 Section 1.5 states that "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety-significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, as endorsed by RG 1.201, with the exception of the seismic hazard impact evaluation, which will use the Electric Power Research Institute (EPRI) 3002017583 (Reference 5) approach for seismic Tier 1 sites. The inclusion of additional process steps to address seismic considerations will ensure that the reasonable confidence required by 10 CFR 50.69(c)(1)(iv) is achieved. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by § 50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence for completing the categorization elements. Therefore, the order in which each element of the categorization process (listed below) is completed is flexible as long as all elements are completed; they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 (seven qualitative criteria) to be completed for components or functions categorized as LSS by all other elements. Similarly, Item 4 (Defense-in-Depth assessment) is only required for safety-related active components or functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and fire PRAs)
2. Non-PRA approaches (e.g., other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. Defense-in-Depth (DID) assessment

5. Passive categorization methodology

Figure 3-1 illustrates the major steps of the categorization process described in NEI 00-04. As explained further in Section 3.2.3, two categorization steps (represented by four blocks on the figure) have been included to highlight review of seismic insights as they pertain to this application.

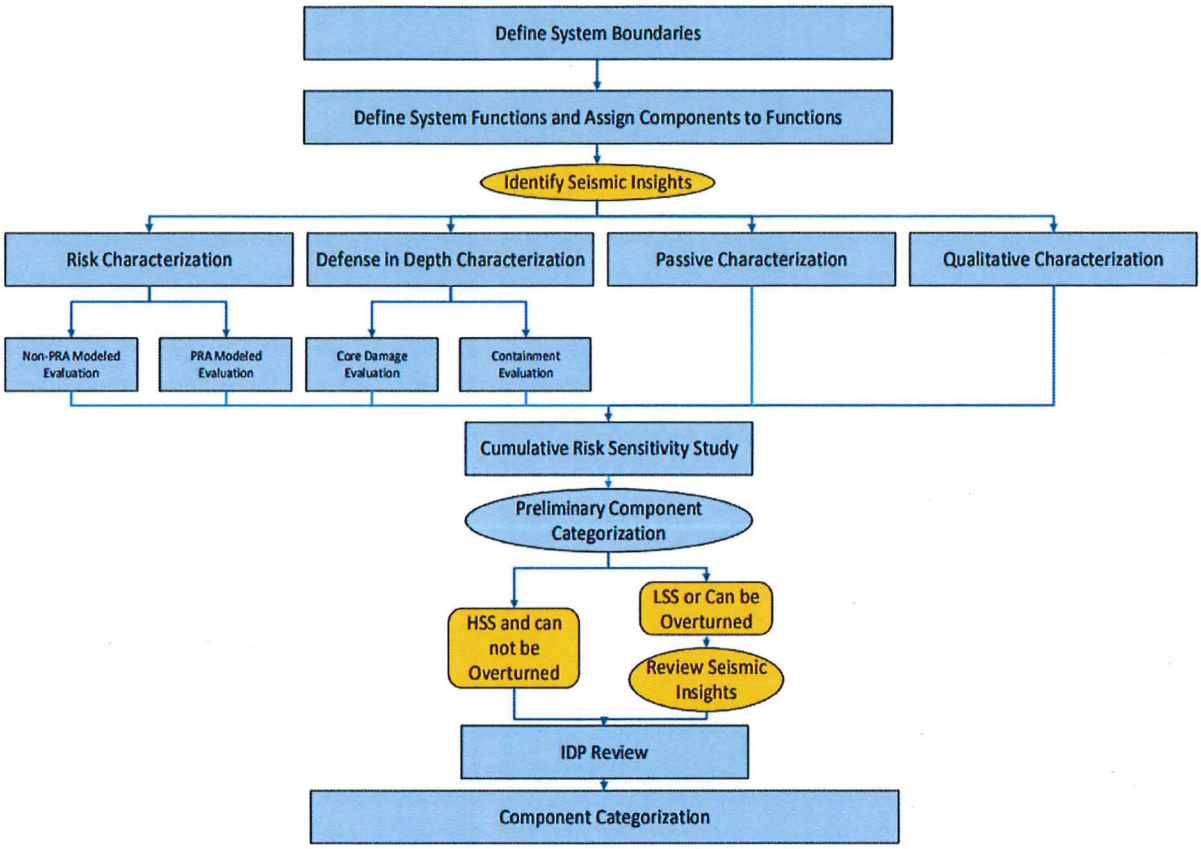

Figure 3-1: Categorization Process Overview

Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., HSS or LSS) that is presented to the Integrated Decision-Making Panel (IDP). Note that the term "preliminary HSS or LSS" used in this application is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1. The safety significance determination of each element is independent and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed

by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final RISC category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04, Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS, however, the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 3-1 summarizes the NEI 00-04 IDP limitations. The steps of the process are performed at either the function level, component level, or both. This is also summarized in Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

Table 3-1: Categorization Evaluation Summary

Element	Categorization Step – NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire and Other External Hazards	Component	Not Allowed	No
	Seismic	Function/Component	Allowed [1]	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable [2]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

Notes:

[1] IDP consideration of seismic insights can also result in an LSS to HSS determination.

[2] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with NEI 00-04, Section 9.2. In some cases, a 10 CFR 50.69 categorization team may provide

preliminary assessments of the seven considerations for the IDP's consideration, however, the final assessments of the seven considerations are the direct responsibility of the IDP.

The seven considerations are addressed preliminarily by the 10 CFR 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 10 CFR 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 10 CFR 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines that any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., internal events PRA or integral PRA assessment) or DID evaluation will be initially treated as HSS. However, NEI 00-04, Section 10.2, allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS. Also, Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with an HSS function, but do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g., Passive, Non-PRA-modeled hazards – see Table 3-1). Except for seismic, these components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if an HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 or may remain LSS. Given that DAEC is a seismic Tier 1 (low seismic hazard) plant as defined in Reference 5, seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

The following clarifications are applied to the NEI 00-04 categorization process:

- The IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member

of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA.

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address, at a minimum, (1) the purpose of the categorization; (2) present treatment requirements for SSCs including requirements for DBEs; (3) PRA fundamentals; (4) details of the plant-specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and (5) the DID philosophy and requirements to maintain this philosophy.
- The decision criteria for the IDP to categorize SSCs as HSS or LSS in accordance with § 50.69(f)(1) will be documented in NEDA procedures.
- Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding HSS and LSS.
- Passive categorization will be performed using the processes described in Section 3.1.2 of this enclosure. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.
- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.
- NEI 00-04, Section 7, requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle Safety Evaluation (Reference 6) which states "...if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the [DID] assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS."
- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS function components to LSS.
- With regard to the criteria that consider whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, NEDA will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.
- NEDA proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3, as discussed in Section 3.2.3 of this enclosure.

The risk analysis to be implemented for each modeled hazard is described below.

- Internal Event Risks: Internal events and internal flooding PRA, which was previously reviewed by the NRC as part of adopting the SFCP (Reference 2).
- Fire Risks: Fire PRA model, which was previously reviewed by the NRC for transitioning to NFPA 805 (Reference 3).

- Seismic Risks: Alternative approach in EPRI 3002017583 (Reference 5) for Tier 1 plants and additional considerations discussed in Section 3.2.3 of this enclosure.
- Other External Risks (e.g., tornados, external floods): Using the Individual Plant Examination of External Events (IPEEE) screening process accepted by NRC safety evaluation dated March 10, 2000 (Reference 7). Attachment 4, External Hazards Screening, summarizes an update of the disposition of each hazard addressed in the IPEEE. The other external hazards were determined to be insignificant contributors to plant risk.
- Low Power and Shutdown Risks: Qualitative DID shutdown model for shutdown configuration risk management based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference 8), which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from the Tier 1 alternative seismic approach to the Tier 2 approach) will not be made without prior NRC approval. The SSC categorization process documentation will include the following elements:

1. Program procedures used in the categorization
2. System functions, identified and categorized with the associated bases
3. Mapping of components to support function(s)
4. PRA model results, including sensitivity studies
5. Hazards analyses, as applicable
6. Passive categorization results and bases
7. Categorization results including all associated bases and RISC classifications
8. Component critical attributes for HSS SSCs
9. Results of periodic reviews and SSC performance evaluations
10. IDP meeting minutes and qualification/training records for the IDP members

3.1.2 Passive Categorization Process

ANO-2 Method – Control Building Chillers Only

NEDA is requesting approval for the use of the ANO-2 approach solely for performing 10 CFR 50.69 categorization of the Control Building Chiller system. To support the restart of DAEC, new commercial-grade chillers are being procured and will be installed to replace the current Control Building Chillers. The new chillers are being installed in anticipation of recategorizing them under the approved 10 CFR 50.69 program using the ANO-2 approach.

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure-retaining function. Passive components and the passive function of active components within the Control Building Chiller system will be evaluated using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference 9 consistent with the related Safety Evaluation (SE) issued by the NRC.

The RI-RRA methodology is a risk-informed safety classification and treatment program for repair and replacement activities for pressure-retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., DID, safety margins) in determining safety significance. Component supports, if categorized, are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

The use of this method for a 10 CFR 50.69 application was previously approved by the NRC as documented in the final SE for Vogtle Units 1 and 2, dated December 17, 2014 (Reference 6). The RI-RRA method, as approved for use at Vogtle for 10 CFR 50.69, does not have any plant-specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, the RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization since this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment.

The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in Reference 9 for the passive categorization of Class 2, Class 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662, as published in RG 1.147 (Reference 10). Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair and replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure-retaining function, as well as supports, will be assigned HSS for passive categorization, which will result in HSS for their risk-informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization are acceptable and appropriate for use at DAEC for 10 CFR 50.69 SSC categorization of the Control Building Chiller system.

Integration of the Enhanced Passive Methodology

Under the proposed change, NEDA is requesting NRC approval to transition to categorizing passive components, and the passive function of active components, in accordance with EPRI Technical Report (TR) 3002033536, "Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components (Revision 1-A)" (Reference 11). This transition will follow the procurement and installation of the Control Building Chillers categorized under the ANO-2 method.

The enhanced methodology for passive categorization requires a plant-level evaluation of the pressure retention function across all systems. In this approach, the pressure boundary function is treated as a distinct system for the purposes of 10 CFR 50.69 categorization and alternative treatment. This treatment, where the boundaries for the pressure boundary system are limited to pressure retention, is consistent with both the language of the regulation and the Statement of Considerations for the final rule. This ensures that all safety-significant functions are properly categorized, and that no other important functions escape evaluation or the appropriate assignment of safety significance. As documented in the SE (Reference 12), the NRC staff

concluded that defining the pressure boundary function as a system for 10 CFR 50.69 categorization complies with the requirements in 10 CFR 50.69(c)(1)(v).

When categorizing a system that includes both active and passive (pressure boundary) components, active components (those with non-pressure-retaining functions) must continue to be evaluated using the existing NEI 00-04 process. This ensures that all safety functions are properly identified and categorized with respect to their safety significance. In some cases, a component may be associated with both active and passive functions. For example, the internals of a safety-related valve may be categorized as HSS for active functions, while the valve body is categorized as LSS for passive functions, resulting in the overall SSC being classified as RISC-1. In such cases, alternative treatments (e.g., alternative repair/replacement activities) may be applied to the LSS pressure-retaining function, since the pressure boundary is categorized RISC-3, while special treatments are maintained for the active HSS functions categorized as RISC-1. Alternative treatments can be applied to the RISC-3 valve body and other pressure boundary piece parts regardless of whether the SSCs performing active functions are categorized. This approach to alternative treatment requirements is consistent with 10 CFR 50.69(d).

While a pressure boundary component may be categorized as HSS or LSS under the enhanced methodology, no change to the special treatment of the active function is permitted from its current safety-related treatment. Alternate treatment of an active function would only be allowed if the guidance in NEI 00-04 for active functions were applied and the safety-related active function were categorized as LSS (i.e., RISC-3) in accordance with the current NEI 00-04 approach.

As described in Section 2.3 of this enclosure, the proposed license condition accommodates the transition from the ANO-2 approach to the enhanced passive methodology for SSCs in the Control Building Chiller system. Once the plant-level evaluation of the pressure-retaining function is completed across all systems using the enhanced method, those categorizations will replace those previously performed under the ANO-2 approach. It is not intended that NEDA will have the option to choose between passive categorization methodologies. Categorization of the Control Building Chiller system using ANO-2 will remain valid until superseded by the categorization performed using the enhanced method, once approved by the IDP.

As part of completing the plant-level System Categorization Document (SCD) for the passive system using the enhanced methodology, NEDA will compare the new categorization results to those completed under the ANO-2 approach for the Control Building Chiller system. For SSCs that were categorized as LSS under ANO-2 but are categorized as HSS using the enhanced methodology, NEDA will evaluate the appropriate course of action. Based on this evaluation, NEDA will restore special treatment requirements for SSCs that change from RISC-3 to RISC-1, consistent with Section 11.2 of NEI 00-04, and will consider enhanced treatment for SSCs whose categorization changes from RISC-4 to RISC-2, as warranted. These activities will be guided by the feedback and adjustment process described in Section 12 of NEI 00-04, which requires evaluation of the impact of updated risk information on the categorization process results.

Once the plant-level evaluation is completed and approved by the IDP, the ANO-2 approach will no longer be used for 10 CFR 50.69 categorization, consistent with the proposed license condition. NEDA will revise the SCD for the Control Building Chiller system to reflect use of and results from the enhanced passive methodology in place of the ANO-2 method and results.

These revisions may be performed during the periodic review process, consistent with Section 12 of NEI 00-04, unless a more timely action is warranted (e.g., change from RISC-3 to RISC-1).

Enhanced Passive Methodology

Passive categorization will follow the applicable guidance in NEI 00-04, supplemented by the steps outlined in Chapter 4 of EPRI TR 3002033536. The enhanced passive methodology contains a defined set of prerequisites, a predetermined set of HSS systems/subsystems, and a plant-specific search to identify additional pressure boundary components to be added to the HSS scope. The process consists of the following five phases:

- Phase 1 – Prerequisites
- Phase 2 – Predetermined HSS passive SSCs
- Phase 3 – Design and plant-specific search for HSS passive SSCs
- Phase 4 – Sensitivity study and IDP review and concurrence
- Phase 5 – Performance monitoring

Phase 1 – Prerequisites

1. PRA Technical Adequacy

   NEDA maintains robust internal events PRAs, including internal flooding, that address the failure of all pressure boundary components at DAEC. Additional information related to PRA technical adequacy is provided in Section 3.2 of this enclosure. Section 3.3 addresses the PRA review process, which confirmed that the DAEC PRA models are of sufficient quality and level of detail to support the categorization process.

   Key assumptions and sources of uncertainty associated with the DAEC PRA models have been reviewed in the context of this application. The results are summarized in Attachment 6 of this enclosure.

2. Integrity Management

   - NEDA implements robust programs to address localized corrosion (e.g., pitting, microbiologically influenced corrosion), following guidance in References 13 through 17.
   - The station maintains a robust flow-accelerated corrosion (FAC) program aligned with the recommendations in Reference 18.
   - Erosion management at NEDA aligns with the recommendations in Reference 19.

3. Protection Measures for Internal Flooding Events

   Prerequisite 3 addresses protective measures for internal flooding events, such as floor drains, flood alarm equipment, and flood barriers. This prerequisite provides that such protective measures should not be categorized as LSS unless additional evaluations have been conducted to show that loss of these measures, or a subset of them, does not

invalidate the HSS determination provided in Section 4.2 of EPRI TR 3002033536. For example, if a submarine door has been credited to prevent a flood from propagating from one flood zone to another, it will be considered HSS unless an evaluation shows that loss of the door will not significantly increase plant risk (i.e., exceed the risk criteria in Criterion 11 of Section 4.2).

4. Reflect the As-Built As-Operated Plant

   NEDA ensures that the PRA models accurately reflect the as-built, as-operated plant by following a process defined in approved procedures. PRA maintenance and updates are discussed in Section 3.2.6 of this enclosure, and the feedback and adjustment process is addressed in Section 3.5. These processes include periodic reviews of PRA inputs and updates to PRA data.

Phase 2 – Predetermined HSS passive SSCs (Criteria 1 through 10)

Section 4.2 of EPRI 3002033536 defines the scope of systems, subsystems, and piping segments that are to be categorized as HSS. Twelve criteria are used to categorize passive components and their associated supports. SSCs meeting any of Criteria 1 through 10 are categorized as preliminary HSS; those that do not meet any of these criteria are categorized as preliminary LSS. Criteria 1 through 10, along with the HSS criteria considerations in EPRI 3002033536, are used to identify the predetermined HSS passive SSCs.

Phase 3 – Design and plant-specific search for HSS passive SSCs (Criteria 11 and 12)

Criterion 11 is uniquely plant-specific and based on quantitative risk metrics from the internal events and internal flooding PRA models. This involves assessing Core Damage Frequency (CDF), Large Early Release Frequency (LERF), Conditional Core Damage Probability (CCDP), and Conditional Large Early Release Probability (CLERP) metrics. Components exceeding these thresholds are categorized as HSS. This ensures identification of plant-specific outliers based on both risk and consequence.

If a pipe segment or a portion of a pipe segment is LSS according to the first 10 criteria, the impact on risk due to its postulated failure is determined using the plant-specific PRA (i.e., through the application of Criterion 11). The NRC staff noted in the SE (Reference 12) that if a passive component is determined to be HSS by Criteria 1 through 10, then there would be no change in categorization by assessing it against Criterion 11. Therefore, the NRC's approval of the methodology was not dependent on applying Criterion 11 to components that are already identified as preliminary HSS.

Criterion 12 provides three options for addressing piping and component supports, including hangers and snubbers: (1) supports may remain uncategorized, (2) if categorized, supports are assigned the same categorization as the highest-ranked piping segment within the piping analytical model in which the support is included, or (3) a combination of restraints or supports such that the LSS piping and associated SSCs attached to the HSS piping are included in the scope up to a boundary point that encompasses at least two supports in each of the three orthogonal directions.

External events are evaluated in accordance with Section 4.3 of EPRI 3002033536. Preliminary HSS/LSS assignments are reviewed and adjusted to reflect the impact of pressure boundary

failure on external event mitigation. The purpose of this review is to confirm and adjust as necessary, that the assignment of HSS/LSS criteria is valid in the context of other hazards (fire, seismic, other hazards).

Phase 4 – Sensitivity study and IDP review and concurrence

The sensitivity study that is conducted as part of the enhanced passive methodology is addressed in Section 3.2.7 of this enclosure.

In addition to the IDP review and approval process outlined in Section 9 of NEI 00-04, the IDP is tasked with confirming that all steps of the enhanced passive categorization methodology have been followed:

- The IDP shall ensure that the prerequisites cited in Section 4.1 are met.
- The IDP shall confirm the assignment of HSS components (from the results of using Criteria 1 through 11) is appropriate.
- The IDP shall confirm that the assignment of HSS criteria is valid in the context of other hazards (fire, seismic, other hazards).

Phase 5 – Performance monitoring

Performance monitoring is addressed in Section 3.5 of this enclosure.

3.2 Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))

The following sections demonstrate that the quality and level of detail of the processes used in the categorization of SSCs are adequate. The PRA models described below have been peer-reviewed, and there are no PRA upgrades that have not been peer reviewed. The PRA models credited in this request are the same PRA models credited in previously approved risk-informed applications for DAEC (References 2 and 3) with routine maintenance updates applied.

3.2.1 Internal Events and Internal Flooding

The DAEC categorization process for the internal events and internal flooding hazard will use the plant-specific PRA model. The NEDA risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for DAEC. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The DAEC categorization process for fire hazards will use a peer-reviewed plant-specific fire PRA model. The internal Fire PRA model was developed consistent with NUREG/CR-6850 and other methods previously accepted by the NRC (Reference 3). The NEDA risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for DAEC. Attachment 2 of this enclosure identifies the applicable fire PRA model.

3.2.3 Seismic Hazards

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards, such as seismic, 10 CFR 50.69(b)(2) allows, and NEI 00-04 summarizes, the use of other methods for determining SSC functional importance in the absence of a quantifiable PRA (such as Seismic Margin Analysis or IPEEE Screening) as part of an integrated, systematic process. For the DAEC seismic hazard assessment, NEDA proposes to use a risk-informed graded approach that meets the requirements of 10 CFR 50.69(b)(2) as an alternative to those listed in NEI 00-04 Sections 1.5 and 5.3. The approach is specified in EPRI 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization" (Reference 5) and includes additional qualitative considerations provided in this section.

DAEC meets the EPRI 3002017583 Tier 1 criteria for a "Low Seismic Hazard/High Seismic Margin" site. The Tier 1 criteria are as follows:

> Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE [Safe Shutdown Earthquake] between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected.

Note: Reference 5 applies to the Tier 1 sites in its entirety except for Sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

The Tier 1 criterion (i.e., basis) in EPRI 3002017583 is a comparison of the ground motion response spectrum (GMRS, derived from the seismic hazard) to the safe shutdown earthquake (SSE, i.e., seismic design basis capability). U.S. nuclear power plants that utilize the 10 CFR 50.69 Seismic Alternative (EPRI 3002017583) will continue to compare GMRS to SSE.

The trial studies in EPRI 3002017583 show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 1 classification and resulting criteria is not that the design basis insights are adequate. Instead, it is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of the EPRI report.

> At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE [full power internal events] PRA and other risk evaluations along with the required [DID] and [IDP] qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 50.69 categorization.

The proposed categorization approach for DAEC is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA (SPRA). For Tier 1 plants, this approach relies on the insights gained from the SPRAs examined in EPRI 3002017583 along with confirmation that the site GMRS is low. The EPRI report demonstrates

that seismic risk is adequately addressed for Tier 1 sites by the results of additional qualitative assessments discussed in this section and existing elements of the categorization process specified in NEI 00-04.

For example, the 10 CFR 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard-specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total CDF contributed by that hazard. For Tier 1 sites, the seismic risk (CDF/LERF) will be low such that seismic hazard risk is unlikely to influence an HSS decision. In applying the EPRI 3002017583 process for Tier 1 sites to the DAEC 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

EPRI 3002017583 recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002017583 for identifying unique seismic insights.

The seismic fragility of an SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041 (Reference 20) provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand. At sites with lower seismic demands such as DAEC, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference 20. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems, providing the technical basis for allowing use of a graded approach for addressing seismic hazard at DAEC.

There are some plant features such as equipment anchorage that have seismic capacities more closely associated with the site-specific seismic demand since those specific features are specifically designed to meet that demand. However, even for these features, the design basis criteria have intended conservatisms that result in significant seismic margins within SSCs. These conservatisms are reflected in key aspects of the seismic design process. The SSCs used in nuclear power plants are intentionally designed using conservative methods and criteria to ensure that they have margins well above the required design bases. Experience has shown that design practices result in margins to realistic seismic capacities of 1.5 or more.

The following provides the basis for establishing Tier 1 criteria in EPRI 3002017583:

a. SSCs for which the inherent seismic capacities are applicable, or which are designed to the plant SSE, will have low probabilities of failure at sites where the peak spectral acceleration of the GMRS < 0.2g or where the GMRS < SSE between 1 and 10 Hz.

b. The low probabilities of failure of individual components would also apply to components considered to have correlated seismic failures.

c. These low probabilities of failure lead to low seismic CDF and LERF estimates, from an absolute risk perspective.

d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing an HSS to LSS due to the 10 CFR 50.69 Integral Assessment provided the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of EPRI 3002017583 showed that it would be unusual, even for moderate hazard plants, to exhibit any unique seismic insights, including those resulting from correlated failures. While it is prudent to perform additional evaluations to identify conditions where correlated failures may occur for Tier 2 sites, for Tier 1 sites such as DAEC, correlation studies would not lead to new seismic insights or affect the site risk results in any significant way. The test case information from EPRI 3002017583, developed by other licensees, including Case Study A (Reference 21), Case Study C (Reference 22), and Case Study D (Reference 23), as well as responses to requests for information and amendments (References 24 through 32) that clarify aspects of these case studies, provide additional supporting bases for this application. Therefore, these case studies, response letters, and amendments are incorporated by reference into this amendment request.

The Tier 1 to Tier 2 threshold as defined in EPRI 3002017583 provides a clear and traceable boundary that can be consistently applied from site to site. Additionally, because the boundary is well defined, if new information is obtained on the site hazard, a site's location within a particular tier can be readily confirmed. In the unlikely event that the DAEC seismic hazard changes to medium risk (i.e., Tier 2) at some future time, NEDA will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e). In accordance with the proposed license condition, NEDA will seek prior NRC approval if it is determined that a process different from the proposed alternative seismic Tier 1 approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69 (e.g., a change from the Tier 1 approach to the Tier 2 approach).

The following provides the basis for concluding that DAEC meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference 33), NEDA submitted a seismic hazard and screening report to the NRC (Reference 34). Figure 3-2 of this enclosure was produced using data submitted by NEDA in Reference 34. Specifically, the DAEC GMRS and SSE were plotted using data from Table 2.4-1 and Table 3.1-1 of Reference 34, respectively. Consistent with NEDA's conclusions, the comparison shows that the DAEC SSE exceeds the GMRS in the 1 to 10 Hz range of the response spectrum. The GMRS peak acceleration is also less than 0.2g. Therefore, the seismic hazard at DAEC meets both Tier 1 screening criteria described in EPRI 3002017583. The rationale for classifying DAEC as a Tier 1 site will be documented and presented to the IDP for each categorized system.

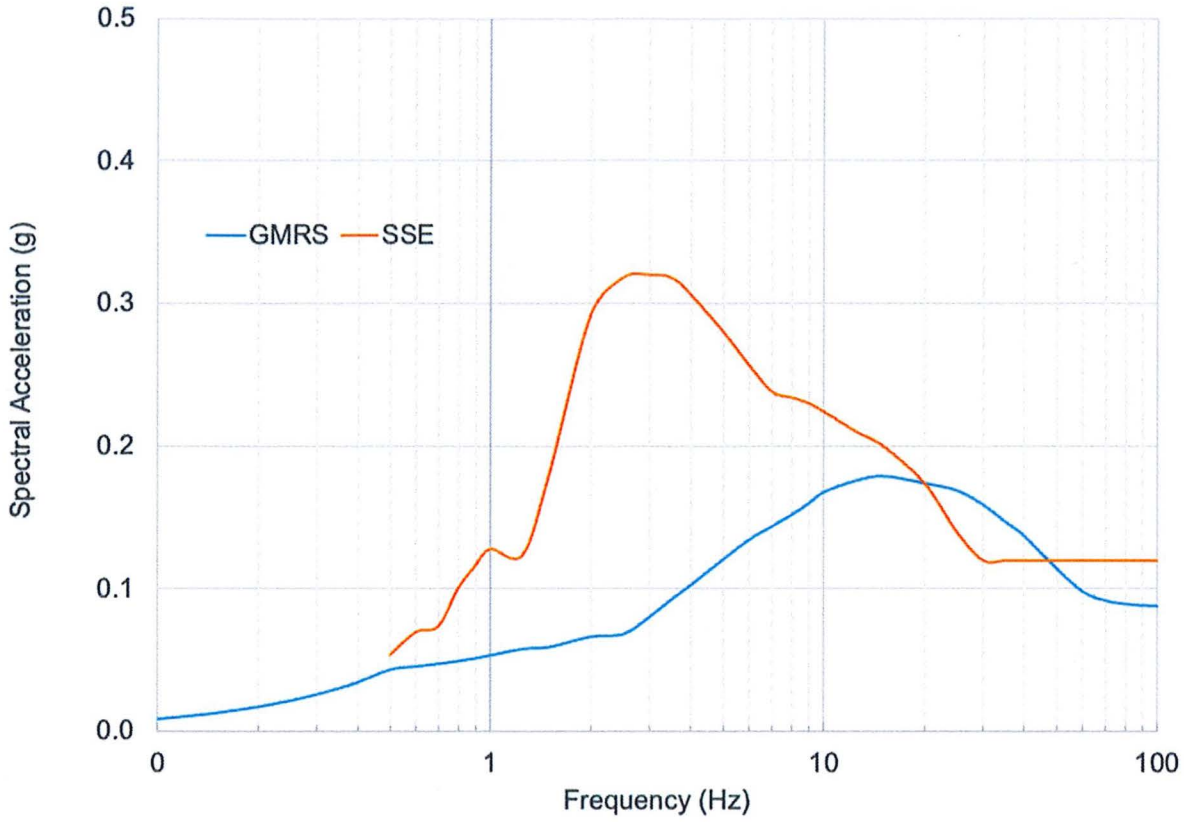

Figure 3-2: Comparison of the DAEC GMRS to the SSE

The NRC staff evaluated NEDA's seismic hazard and screening response for DAEC and performed confirmatory calculations to validate both the methodology and results (Reference 35). The NRC concluded that the methodology employed to determine the GMRS was acceptable and that NEDA adequately characterized the reevaluated seismic hazard for the DAEC site. The NRC staff confirmed that the DAEC site is bounded by the SSE between 1 and 10 Hz of the response spectrum, which supports NEDA's conclusion that DAEC meets the Tier 1 criteria described in EPRI 3002017583.

Section 1.1.3 of EPRI 3002017583 cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For DAEC, the principal seismic reviews prepared by NEDA and the NRC's corresponding regulatory assessments of those reviews are provided in the following licensing documents.

1. Near-Term Task Force (NTTF) Recommendation 2.1 seismic hazard screening (References 34 and 35)
2. NTTF Recommendation 2.3 seismic walkdowns (References 36, 37, 38, and 39)
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (References 40 and 41)

In addition, NEDA submitted a limited scope high frequency evaluation (Reference 42) and the NRC's review of that evaluation confirmed that DAEC met the low spectral acceleration screening criterion (i.e., horizontal GMRS high frequency peak spectral acceleration less than

0.2g). The NRC staff concluded that no further response or regulatory actions associated with the 50.54(f) letter review of Phase 2 of NTTF Recommendation 2.1, "Seismic," were required for DAEC (Reference 43).

DAEC meets the criteria for a Tier 1 (low seismic hazard) site based on the Tier 1 definition in EPRI 3002017583. The DAEC GMRS is below 0.2g across the entire spectral acceleration range, and the SSE bounds the GMRS between 1 and 10 Hz, as documented in Reference 34. Using the conservative seismic risk estimation approach applied by the NRC for DAEC in the GI-199 evaluation, in which the plant-level seismic High Confidence Low Probability of Failure (HCLPF) is conservatively assumed to equal the DAEC SSE of 0.12g peak ground acceleration (PGA), and using the DAEC PGA seismic hazard curve from the 2014 NTTF 2.1 Seismic Hazard Screening submittal, the DAEC seismic core damage frequency (SCDF) is estimated to be approximately 5E-6/yr. Therefore, consistent with the EPRI Tier 1 treatment, unique seismic categorization insights are not expected, and seismic considerations are not expected to result in an SSC being designated HSS that would otherwise be classified LSS based on the other categorization inputs.

As an enhancement to the EPRI study results as they pertain to DAEC, the proposed categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for DAEC. For example, the SCD that is presented to the IDP will include a summary of the identified plant seismic insights pertinent to the system being categorized. It will also state the basis for applicability of the EPRI 3002017583 study as well as the bases for DAEC being a Tier 1 plant. The discussion of the Tier 1 bases will include such factors as:

- The low seismic hazard for the plant, which is subject to periodic reconsideration as new information becomes available through industry evaluations; and
- The definition of Tier 1 in the EPRI study.

At several steps of the categorization process (e.g., as noted in Figure 3-1 and Table 3-1 of this enclosure) the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. Integrated importance measures over all the modeled hazards (i.e., internal events, internal flooding, and internal fire for DAEC) are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS. HSS SSCs uniquely identified by the DAEC PRA models also having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

For components that are HSS due to fire PRA but not HSS due to internal events PRA, the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation to the IDP as additional qualitative inputs described in the SCD.

The categorization team will review available DAEC plant-specific seismic reviews and other resources such as those identified above. The objective is to identify plant-specific seismic

insights relevant to the components in the system being categorized, that might include potentially important impacts such as:

- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components implicitly part of PRA-modeled functions (including relays)
- Components that may be subject to correlated failures

These impacts would be compiled on an SSC basis. As each system is categorized, the system-specific seismic insights will be provided to the IDP as part of the review process. The IDP can challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if they believe there is a basis for doing so. Any decision by the IDP to downgrade preliminary HSS components to LSS will also consider the applicable seismic insights in that decision. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

Use of the EPRI approach outlined in Reference 5 to assess seismic hazard risk for 10 CFR 50.69 with the additional reviews discussed above will provide a process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs that satisfies the requirements of 10 CFR 50.69(c).

Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference 5 applies to DAEC, i.e., DAEC is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference 5, the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. The Calvert Cliffs responses to requests for additional information (References 44, 45, and 46) are incorporated by reference into this amendment request because they provide additional supporting bases for Tier 1 plants that are applicable to DAEC. Therefore, with little to no anticipated unique seismic insights, the 10 CFR 50.69 categorization process using the full power internal events (FPIE) PRA and other risk evaluations, along with the IDP's DID and qualitative assessment, effectively identifies the safety-significant functions and SSCs.

3.2.4 Other External Hazards

All other external hazards, except for seismic, were screened for applicability to DAEC using a plant-specific evaluation in accordance with Generic Letter 88-20 (Reference 47) and updated to use the criteria in ASME PRA Standard RA-Sa-2009 (Reference 48). Attachment 4 of this enclosure provides a summary of the other external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

As part of the categorization assessment of other external hazards, an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS and cannot be overridden by the IDP. Table 3-2 provides the flood protection features relied upon for screening the Local Intense Precipitation (LIP) flooding mechanism. Table 3-3 provides the flood

protection features relied upon for screening the Flooding in Streams and Rivers mechanism; the credited flood protection strategy includes additional flood barriers that are installed prior to flood waters arriving on site. The SSCs in Table 3-2 and Table 3-3 shall be categorized HSS if the flood protection systems are included in the scope of 10 CFR 50.69 categorization.

Table 3-2: Flood Protection Features Credited for Screening LIP

Door Number	Location	Description
Door 124	Turbine Building	Permanent watertight door
Door 136	Turbine Building	Stairwell 14 to yard door
Door 137	South Turbine Building	Roll up door
Door 154	North Turbine Building	Yard walkout door

Table 3-3: Flood Protection Features Credited for Screening River Flooding

#	Building / Area	Doors / Features Requiring Flood Protection	Flood Protection Function
1	Reactor Building	Door No. 225 – Access Control Airlock Door	Supports installation of credited flood barrier/stoplogs
2	Reactor Building	Door No. 239 – Railroad Airlock Door	Supports installation of credited flood barrier/stoplogs
3	Reactor Building / Turbine Building Interface	Door Nos. 229, 230, and 127 / associated structural interface	Maintains credited flood boundary when temporary protection is installed
4	Turbine Building	Door No. 124 – North Turbine Building Rollup Door / Watertight Floodgate 124B	Provides credited flood protection
5	Turbine Building	Door No. 136 – Stairwell 14 to Yard Door	Supports installation of credited flood barrier/stoplogs
6	Turbine Building	Door No. 137 – South Turbine Building Rollup Door	Supports installation of credited flood barrier/stoplogs
7	Turbine Building	Door No. 154 – North Turbine Building to Yard Walkout Door	Supports installation of credited flood barrier/stoplogs
8	Turbine Building / Control Building Interface	Door Nos. 121 and 122 / associated structural interface	Maintains credited flood boundary when temporary protection is installed
9	Control Building	Door No. 421 – Battery Corridor to Administration Building	Supports installation of credited flood barrier/stoplogs
10	Pump House	Door No. 500 – Pump House Entry Door	Supports installation of credited flood barrier/stoplogs

#	Building / Area	Doors / Features Requiring Flood Protection	Flood Protection Function
11	Pump House	Door No. 507 – Circulating Water Pump Area to RHRSW Pump Area	Provides credited internal flood boundary when closed
12	Pump House	Floor Hatches PR-7, PR-8, and PR-9	Provides credited flood boundary
13	Recombiner Building	Door No. 243 – Offgas/Recombiner Building to Yard Door	Supports installation of credited flood barrier/stoplogs
14	LLRPSF	Door Nos. 805, 806, and 846	Support installation of credited flood barriers/stoplogs
LLRPSF – Low-level Radioactive Waste Processing and Storage Facility			

3.2.5 Low Power and Shutdown

Consistent with NEI 00-04, the NEDA categorization process will use the shutdown safety management plan described in NUMARC 91-06 (Reference 8) for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet either of the two criteria (i.e., considered part of a "primary shutdown safety system" or a failure would initiate an event during shutdown conditions) described in Section 5.5 of NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

The NEDA risk management process ensures that the applicable PRA models used in this application continue to reflect the as-built and as-operated plant. The process delineates the responsibilities and guidelines for updating the PRA models and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be reevaluated.

In addition, NEDA will implement a process that addresses the requirements in NEI 00-04, Section 11, "Program Documentation and Change Control." The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

The DAEC PRA model was last published shortly before the decision was made to permanently cease power operations. At that time, DAEC had transitioned to NFPA 805, and the associated fire protection changes had been incorporated into the PRA model. However, implementation of certain other plant modifications, including the hardened wetwell vent, had not yet been completed.

Following the shutdown decision, development and maintenance of the operating plant PRA model were suspended because no further changes to the design or operation of systems credited in the PRA were being implemented for continued power operation. As part of the current restart effort, plant modifications and equipment restoration activities are being processed through the station design change and fire protection program review processes. These processes identify the PRA organization as a stakeholder, such that changes with the potential to affect the PRA are evaluated and incorporated, as appropriate, into the PRA model.

These processes will support development of the final PRA model used to support 10 CFR 50.69 categorization so that it reflects the as-built, as-operated DAEC configuration associated with resumed power operation.

3.2.7 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04.

In the overall risk sensitivity studies, NEDA will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference 6. Consistent with the NEI 00-04 guidance, NEDA will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

Sensitivity studies conducted as part of the enhanced passive categorization methodology will follow the guidance in Section 4.4 of EPRI 3002033536. For pressure boundary components modeled in the internal events or internal flooding PRA and initially determined to be LSS, their failure rates (e.g., pipe break frequency) are increased by a factor of 3. The CDF and LERF values are then recalculated and compared against the quantitative acceptance guidelines in RG 1.174. Pressure boundary components modeled in the full-power internal events PRA that exceed the RG 1.174 acceptance guidelines will be considered candidate HSS, subject to concurrence by the IDP. Conversely, pressure boundary components determined to be LSS per

Section 4.2 of EPRI 3002033536, and whose sensitivity study results remain below the RG 1.174 acceptance guidelines, will continue to be categorized as LSS.

Additional information is provided in Attachment 6, including a description of the DAEC PRA model key assumptions and sources of uncertainty and their impact on 10 CFR 50.69 categorization.

3.3 PRA Review Process Results (10 CFR 50.69(b)(2)(iii))

The PRA models described in Section 3.2 have been assessed against the applicable ASME/ANS PRA Standard requirements, as endorsed and clarified by RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," consistent with NRC RIS 2007-06 (Reference 49).

The DAEC FPIE PRA model was subjected to a peer review in 2007 using ASME RA-Sb-2005, as endorsed and clarified by RG 1.200, Revision 1 (Reference 50). A focused-scope peer review was subsequently performed in March 2011 using the NEI 05-04 process and ASME/ANS RA-Sa-2009 (Reference 48), as endorsed and clarified by RG 1.200, Revision 2 (Reference 51). The 2011 focused-scope review evaluated the disposition of findings and suggestions from the 2007 peer review and PRA upgrades made since that review, including applicable internal flooding technical elements, and addressed the internal events at-power PRA technical elements.

The DAEC fire PRA was subjected to a peer review in November 2010 against the applicable requirements of ASME/ANS RA-Sa-2009. The fire PRA peer review resulted in finding-level facts and observations (F&Os) that were subsequently addressed through PRA model and documentation updates. The 2017 DAEC peer review finding resolution documentation identifies the November 2010 fire PRA peer review and the March 2011 focused-scope internal events peer review as the sources of the F&Os carried forward for independent review.

In 2017, an independent F&O closure review was performed for outstanding F&Os from the internal events and fire PRA peer reviews in accordance with the Appendix X process, "Close-Out of Facts and Observations (F&Os)" (Reference 53), and the applicable requirements of ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2. The review concluded that none of the changes made to address the F&Os constituted a PRA upgrade or introduced a new PRA method.

In August 2026, an additional independent assessment of the remaining F&Os was performed in accordance with NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standards," and the applicable peer review provisions of RG 1.200, Revision 3 (Reference 54). The assessment reviewed F&Os from the FPIE and fire PRA peer review history. Of the F&Os reviewed, only two remain open. The independent assessment also concluded that none of the F&O resolutions reviewed constituted a PRA upgrade.

The two remaining open F&Os, 5-27 and 5-30, are associated with fire PRA Supporting Requirement HRA-A4 and concern the need for detailed reviews with Operations and/or Training to confirm that significant fire PRA scenarios and associated human failure events are consistent with plant operating and training practices. Attachment 3 provides the disposition of these two remaining open F&Os and their impact on this application.

The peer review and independent assessment results, together with the disposition of the remaining open F&Os provided in Attachment 3, demonstrate that the DAEC PRA models are of sufficient technical adequacy and level of detail to support the categorization process. Additionally, each applicable PRA model has been subjected to a peer review process assessed against an NRC-endorsed standard or set of acceptance criteria, as required by 10 CFR 50.69(c)(1)(i).

3.4 Risk Evaluations (10 CFR 50.69(b)(2)(iv))

The NEDA 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of § 50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04, Section 8, will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF. The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 Feedback and Adjustment Process

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, a timely evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle.

The operational programs required by the Enhanced Passive Risk-Informed Categorization Methodology for Pressure Boundary Components provide robust performance monitoring of passive SSCs susceptible to localized corrosion, including flow-accelerated corrosion (FAC) and erosion. The plant's performance monitoring process is designed to detect changes in component performance, ensuring that potential increases in failure rates of categorized components are identified and addressed before they exceed the rates assumed in the sensitivity study. Because passive components typically exhibit low failure rates, even a single failure event can significantly alter the failure frequency for a specific type of piping, potentially exceeding acceptable guidelines for that subset of components. This underscores the importance of maintaining robust integrity management programs, as required by Prerequisite 2 of EPRI 3002033536 and 10 CFR 50.69(e). If any of these integrity management programs are discontinued or modified in such a way that Prerequisite 2 is no longer satisfied, the enhanced passive categorization methodology will no longer be applied. Future changes to these integrity management programs must be evaluated under 10 CFR 50.69(e) to ensure continued compliance with the methodology.

To more specifically address the feedback and adjustment process (i.e., performance monitoring) as it pertains to the proposed DAEC Tier 1 approach discussed in Section 3.2.3 of this enclosure, implementation of the NEDA design control and corrective action programs provide assurance that the inputs for the qualitative determinations for seismic continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

The performance monitoring process will be described in NEDA's 10 CFR 50.69 program documents. The program requires that the periodic review assess changes that could impact the categorization results and provides the IDP with an opportunity to recommend categorization and treatment adjustments. Personnel from engineering, operations, PRA group, regulatory affairs, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability; to identify and reverse negative performance trends and take corrective action, if necessary.

The NEDA configuration control process ensures that changes to the plant, including a physical change to the plant and/or changes to documents, are evaluated to determine the impact to drawings, design bases, licensing documents, programs, procedures, and training. The configuration control program will include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69 to ensure that physical changes to the plant or changes to plant documents are evaluated.

The checklist will include:

- A review of the impact on the SCD for configuration changes that may impact a categorized system under 10 CFR 50.69.
- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures.
- Review of impact to seismic loading and SSE seismic requirements, as well as the method of combining seismic components.
- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

NEDA has a comprehensive problem identification and corrective action program that ensures that issues are identified and resolved. Issues that may impact the 10 CFR 50.69 categorization process will be identified and addressed through the problem identification and corrective action program, including seismic-related issues.

The NEDA 10 CFR 50.69 program requires that system categorization cannot be approved by the IDP until the panel's comments have been resolved to the satisfaction of the IDP. This includes issues related to system-specific seismic insights considered by the IDP during categorization.

Scheduled periodic reviews are completed at least once every two refueling cycles and will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This scheduled review will include:

- A review of plant modifications since the last review that could impact the SSC categorization.

- A review of plant-specific operating experience that could impact the SSC categorization.
- A review of the impact of the updated risk information on the categorization process results.
- A review of the importance measures used for screening in the categorization process.
- An update of the risk sensitivity study performed for the categorization.

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded, a review of the SSC categorization will be performed.

The periodic monitoring requirements of the 10 CFR 50.69 process will capture these issues and address them at a frequency commensurate with the issue severity. The 10 CFR 50.69 periodic monitoring program includes immediate and periodic reviews, that include the requirements of the regulation, to provide assurance that issues that could affect 10 CFR 50.69 categorization are addressed. The periodic monitoring process also monitors the performance and condition of categorized SSCs such that the assumptions for reliability in the categorization process are maintained.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."
- NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 No Significant Hazards Consideration Analysis

NextEra Energy Duane Arnold, LLC (NEDA) proposes to modify the licensing basis for Duane Arnold Energy Center (DAEC), to allow for the voluntary implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

NEDA has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of structures, systems and components (SSCs) subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment

requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NEDA concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed

manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

A review has determined the proposed change qualifies for a categorical exclusion from environmental review in accordance with 10 CFR 51.22. The proposed action is an amendment regarding the installation or use of a facility component as described in 10 CFR 51.22(d)(8). The proposed action does not disturb any previously undisturbed ground and there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, does not result in a significant increase in individual or cumulative public or occupational radiation exposure, and does not result in a significant increase in the potential for or consequences from radiological accidents. Therefore, pursuant to 10 CFR 51.22, neither an environmental assessment nor an environmental impact statement is required.

6. REFERENCES

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3. NRC Letter to DAEC, "Duane Arnold Energy Center – Issuance of Amendment Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c)," September 10, 2013 (ML13210A449)
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Attachment 1: List of Categorization Prerequisites

NextEra Energy has established fleet procedures that outline the process for performing 10 CFR 50.69 categorization. These fleet procedures will be implemented at DAEC prior to performing system categorizations on DAEC systems. The fleet procedures to be implemented at DAEC contain the elements/steps listed below.

- Integrated Decision-Making Panel (IDP) member qualification requirements.
- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04. Any component supporting an HSS function is categorized as preliminary HSS. Components supporting an LSS function are categorized as preliminary LSS.
- Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.
- Assessment of defense-in-depth (DID) and safety margin. Safety-related components that are categorized as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.
- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.
- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.
- Complete the Operations and/or Training reviews identified in Fire PRA Findings 5-27 and 5-30 and update the associated documentation.
- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.
- Documentation requirements per Section 3.1.1 of the enclosure.

Attachment 2: Description of PRA Models Used in Categorization

Model	Baseline CDF (/yr)	Baseline LERF (/yr)	Comment
Internal Events and Internal Flooding	2.55E-06	7.80E-07	The Internal Events and Internal Flooding PRA model used for this application is the PDA26A application-specific model documented in PDA-PRA-ASM-PDA26A, Revision 0.
Fire	1.31E-05	5.59E-06	The Fire PRA model used for this application is the PDA26AF application-specific model documented in PDA-PRA-ASM-PDA26AF, Revision 0.

Attachment 3: Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
5-27	HRA-A4	NOT MET	The peer review determined that, although the fire PRA documentation indicated that operator interviews had been conducted, adequate documentation of the interviews was not available and relatively few fire-specific operator insights were incorporated into the HRA. Operator talk-throughs or reviews are needed to confirm that interpretation of the procedures and modeled human failure events are consistent with plant operating and training practices.	Prior to implementation of the 10 CFR 50.69 categorization process, detailed reviews with Operations and/or Training will be performed for significant fire PRA scenarios to confirm that the applicable procedures, sequence of events, cues, timing, performance-shaping factors, and modeled human failure events are consistent with plant operating and training practices. The fire PRA documentation will be updated, as appropriate, to incorporate the results.
5-30	HRA-A4	NOT MET	The peer review determined that simulator observations or detailed operator talk-throughs had not been performed to confirm the response models for fire scenarios. Review of significant fire scenarios with Operations is needed to confirm that the modeled sequence of events and associated human failure events are consistent with actual operator response and training.	

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	C2 PS4	DAEC satisfies Criterion 1.A of NRC Standard Review Plan (SRP) 3.5.1.6 (ML100331298). The nearest airfield used for commercial traffic is the Cedar Rapids Municipal Airport, located 15 miles south-southeast of the site. Annual operations (calendar year 2025) from this airport are less than 45,000, which is much less than the 1000 D^2 criterion of approximately 225,000 flight operations per year.
Avalanche	Y	C3	Excluded due to site topography that would not support snow buildup that would lead to an avalanche.
Biological Event (Animal Infestation)	Y	C4 C5	The hazard is included implicitly in the loss of offsite power (LOOP) initiator. These are slow-developing events and have limited impact. Preventative measures are in place through station housekeeping and material control procedures.
Biological Event (Aquatic Growth)	Y	C1 C5	DAEC evaluated the potential for intrusion of biological fouling organisms and conducts semiannual sampling of the intake structure and cooling tower basin to detect possible future intrusion. Plant procedures establish the strategic chemistry plan, including microbiological and macrobiological fouling control programs and describe monitoring and control approaches, including chemical treatment, sampling, and inspections. Aquatic growth is a gradual and detectable condition that is managed through intake monitoring, sampling, inspection, chemical treatment, and open cooling water chemistry controls.
Biological Event (Organic Material in Water)	Y	C1 C5	River sediment and debris, including deadwood, are addressed through periodic river surveys, dredging, and removal of deadwood and tree trunks upstream of the intake. Water entering the intake structure passes through bar racks and traveling screens before reaching the river water pump wet pits, and a trash rake is provided to remove debris accumulated on the bar racks. These design and operational provisions provide

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			for detection and removal of river-borne debris before it can significantly impair required cooling-water flow.
Coastal Erosion	Y	C1 C3	DAEC is located inland along the Cedar River and not on a coastline, ocean, Great Lake shoreline, or tidal water body subject to coastal processes. The site is founded on competent bedrock and has engineered riverbank protection features.
Drought	Y	C1 C5	Drought is a slow-developing event for which adequate time would be available to safely shut down the plant.
External Flooding	Y	C1	The external flooding mechanisms were evaluated for the 10 CFR 50.69 categorization process. Only Local Intense Precipitation (LIP) and Flooding in Streams and Rivers mechanisms were determined to not be bounded in the DAEC Flood Hazard Reevaluation Report (FHRR) (ML14072A019, ML14072A020). All other flood hazard mechanisms are bounded by the plant design. The evaluations concluded that no key structures, systems, or components (SSCs) are impacted by LIP due to the adequate Available Physical Margin (APM) provided by the existing plant configuration and that adequate APM exists in the licensing basis flood response features to protect all key SSCs from Flooding in Streams and Rivers. Therefore, external flooding mechanisms can be screened from further consideration in the 10 CFR 50.69 categorization process given that the Flood Protection SSCs are designated as high safety significant (HSS) for 10 CFR 50.69. These SSCs are discussed in Section 3.2.4 of the enclosure.
Extreme Wind or Tornado	Y	C1 PS4	The DAEC design basis provides protection against extreme wind and tornado effects, including tornado-generated missiles. The frequency of wind-pressure effects exceeding the design basis is less than 1E-06/yr. Tornado missile protection is provided through the plant licensing-basis structural barriers, equipment location, and separation provisions. The June 2026 tornado missile walkdown confirmed that the observed plant configuration

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			remains generally consistent with the licensing basis and prior plant-specific evaluations, with no new common-cause tornado missile vulnerability identified. In addition, the IPEEE estimated the bounding core damage frequency (CDF) contribution from high winds and tornadoes to be approximately 1.41E-7/yr, which is below the 1E-6/yr progressive screening criterion. This screening basis is contingent upon restoration of the Control Room east-wall opening and any other temporarily removed barriers or SSCs to their approved design configuration prior to plant startup.
Fog	Y	C4	The principal impacts of fog would be indirect effects such as transportation-related accidents or weather-related LOOP events. These contributors are implicitly included within industry operating experience and initiating event frequencies represented in the internal events PRA model. See Transportation Accidents.
Forest Fires	Y	C1 C4	The site is not located in a heavily forested region subject to significant wildfire exposure. The effects of forest or range fires are implicitly included in the PRA LOOP initiator.
Frost	Y	C1 C4	Frost effects are encompassed by the more severe events of snow and ice storms. The effects of frost are included implicitly in the PRA LOOP initiator. See Snow and Ice Cover hazards.
Hail	Y	C1 C4	The possibility of hailstorms and their effects have been considered in the plant design basis. The effects of hail are included implicitly in the LOOP initiator.
High Summer Temperature (Air)	Y	C1 C5	The plant's heating, ventilation, and air conditioning (HVAC) systems are designed such that abnormal outside ambient temperatures will not impair equipment operation, and redundant HVAC systems are provided for safety-related areas. High summer temperature is a gradual and forecastable condition. Plant procedures include actions for forecasted extreme hot weather conditions and provide for monitoring and response to elevated ambient temperatures before safety-related equipment is challenged.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
High Summer Temperature (Water)	Y	C1 C5	River water temperature is monitored at the intake structure, and the plant is designed to maintain the required emergency cooling capability during expected seasonal river conditions. High river water temperature is a gradual and forecastable condition that provides substantial operator response time. In addition, Technical Specification (TS) 3.7.2 includes surveillance requirements (SRs) to verify that average river water temperature remains ≤ 95°F. Plant procedures include actions for forecasted extreme hot weather conditions and provide additional monitoring and operational response capability before cooling water performance could be adversely affected.
High Tide, Lake Level, or River Stage	Y	C1 C5	DAEC is not located near a lake, and lake level is not applicable to the site as a hazard. The plant design basis includes consideration of extreme river level and flood conditions, and flood protection features are addressed within the current licensing basis. In addition, TS 3.7.2 includes SRs to verify river water level and river water depth at the intake structure. See also External Flooding.
Hurricane	Y	C3	Not applicable to DAEC since the plant is not located in a hurricane zone.
Ice Cover	Y	C1 C4 C5	Ice blockage resulting in critical low flow is not considered credible for more than a brief period, and the plant incorporates intake monitoring, river management, and operational controls to maintain required cooling water capability during winter conditions. The principal risk-significant effects of this hazard are indirect and are represented in the internal events PRA through the LOOP initiating event frequency. Plant procedures require cold weather preparations and increased monitoring of intake-related and other vulnerable plant areas during severe winter weather conditions.
Industrial or Military Facility Accident	Y	C1 C3	There are no nearby military facilities. Nearby industrial, transportation, and hazardous chemical facilities have been evaluated for toxic gas releases, fire hazards, explosions, and nearby pipeline hazards. Nearby hazardous chemical storage, including anhydrous ammonia, does not

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			present a threat to control room habitability, and surrounding gas pipelines present negligible risk to safe plant operation.
Internal Flooding	N	None	The internal flooding PRA model addresses risks from internal flood events.
Internal Fire	N	None	The internal fire PRA model addresses risks from internal fires.
Landslide	Y	C3	Excluded due to site topography that would not support landslide of any significance.
Lightning	Y	C4	The principal effects of lightning would be indirect impacts such as transmission disturbances, switchyard faults, or weather-related LOOP events. These contributors are represented within the internal events PRA initiating event frequencies.
Low Lake Level or River Stage	Y	C1 C5	Low river level and intake operability are addressed through intake flow management and river maintenance features, including dredging, guide walls, Iowa vanes, spur dikes, riprap protection, sediment management, and periodic river surveys. In addition, TS 3.7.2 includes SRs to verify that river water level and river water depth meet minimum requirements. Low river stage is a gradual condition with substantial warning time and is managed through normal operational monitoring and response actions. There are also no man-made structural features that could result in a loss of the Cedar River as a heat sink.
Low Winter Temperature (Air)	Y	C1 C5	DAEC is designed for sustained winter weather conditions typical of the site region. Seasonal cold weather conditions are gradual and predictable. Plant procedures require verification of cold weather preparations, monitoring of outside temperature and wind conditions, increased monitoring of vulnerable plant areas, and implementation of protective actions during forecasted extreme cold weather conditions. These established operational and seasonal readiness measures provide significant time to identify and mitigate adverse impacts before safety-related equipment is challenged.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Low Winter Temperature (Water)	Y	C1 C5	Ice blockage resulting in critical low flow is not considered credible for more than a brief period and adequate river flow is available to support required cooling water functions. Low winter river temperature and ice-related conditions develop gradually and are managed through normal operational activities. Plant procedures require verification of cold weather preparations and increased monitoring of intake-related and other vulnerable plant areas during forecasted extreme cold weather conditions, providing additional opportunity to identify and respond to developing conditions before intake operability is affected.
Meteorite or Satellite Impact	Y	PS4	The hazard is screened based on low hazard frequency, consistent with NUREG-1407 (ML063550238).
Pipeline Accident	Y	C3	Pipelines in the vicinity of DAEC have been evaluated for potential pipeline rupture effects, including hazardous material release and explosion/fire effects. The pipelines are sufficiently small and distant and present negligible risk to safe operation of the plant.
Release of Chemicals in Onsite Storage	Y	C1 C3	Onsite chemical inventories have been evaluated and do not represent a threat to control room habitability. The onsite 1,500-gallon liquid oxygen tank satisfies applicable separation criteria. A surveillance performed in 2020 documented no hazardous chemical conditions requiring additional action or presenting a challenge to control room habitability.
River Diversion	Y	C1 C5	River diversion is a sufficiently slow-developing event such that adequate time would be available to safely shut down the plant. There is an extremely low probability that the Cedar River will fail to provide sufficient cooling water; the intake structure and river water supply features are designed to maintain the required cooling water capability. Failure of the sediment management system, including the Iowa Vanes and steel sheet guide wall, would have no adverse impact on the intake structure or river water supply system.
Sand or Dust Storm	Y	C3	DAEC is located in an inland river environment that is not subject to desert or coastal sandstorm conditions, and there are no nearby geographic

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			features or large loose sand sources capable of generating significant sand or dust storm hazards.
Seiche	Y	C3	Not applicable to DAEC per the FHRR.
Seismic Activity	N	None	DAEC is a Tier 1 Plant as defined by EPRI 3002017583. See Section 3.2.3 of the enclosure.
Snow	Y	C1 C4 C5	Plant structures are designed for a roof snow load of 40 psf. The principal risk-significant effects of snowfall are indirect impacts such as transmission disturbances or weather-related LOOP events, which are represented within the internal events PRA initiating event frequencies. Significant snowfall is forecastable and develops with warning time, allowing implementation of seasonal readiness and response actions. Plant procedures require verification of cold weather preparations, monitoring of weather conditions, increased monitoring of vulnerable plant areas, and implementation of protective measures during severe winter weather conditions.
Soil Shrink-Swell Consolidation	Y	C1 C5	Soil settling has been considered in the design of the plant. In addition, soil shrink-swell and consolidation are sufficiently slow-developing events that allow adequate time for safe shutdown of the plant. The site is founded on competent bedrock and engineered structural fill and is not subject to significant shrink-swell soil behavior or consolidation mechanisms capable of challenging safety-related SSCs.
Storm Surge	Y	C3	Not applicable to DAEC per the FHRR.
Toxic Gas	Y	C1 C3	Onsite and offsite hazardous chemical storage and potential toxic gas release hazards within the vicinity of DAEC have been evaluated. The original onsite chlorine gas hazard was eliminated through replacement of chlorine gas storage with sodium hypochlorite treatment systems. Offsite chemical surveys within five miles identified no additional chemicals presenting a hazard to control room habitability; nearby anhydrous ammonia storage does not present a threat to safe plant operation or control room habitability. A surveillance performed in May 2020 documented no hazardous chemical conditions requiring additional action or presenting a challenge to control room habitability.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Transportation Accident	Y	C1 C2 C3 C4	Relevant hazards in the vicinity of DAEC have been evaluated, including highways, railroads, waterways, airports, pipelines, and other hazardous material transportation routes and nearby facilities. Transportation hazards involving marine, railroad, and truck accidents were determined to present no significant risk to the plant based on separation distance. Pipeline hazards were found to present no significant risk due to their size and distance from the site. Evaluations concluded that transportation hazards do not pose a significant threat to plant safety and that the contribution to CDF is below the NUREG-1407 screening criterion. See also Aircraft Impacts.
Tsunami	Y	C3	Not applicable to the DAEC per the FHRR.
Turbine-Generated Missiles	Y	PS4	Turbine-generated missile hazards have been evaluated, considering turbine wheel failures, missile trajectories, penetration capability, and potential impacts on safe-shutdown equipment. The FSAR identifies a total 40-year turbine wheel failure probability of 5.6E-07 and an average annual wheel failure probability of 1.4E-08. In addition, the reactor building and control building walls and slabs are designed to withstand turbine-generated missiles, and all equipment required for safe shutdown was reviewed for vulnerability to postulated missile effects.
Volcanic Activity	Y	C3	Excluded due to site topography. There are no active volcanoes in the vicinity of DAEC.
Waves	Y	C1	A combined event was evaluated that consists of a 100-year snowpack, snow-season maximum rainfall, and waves generated by a 2-year wind event occurring at the peak still-water level. The resulting maximum flood elevations include wind-wave run-up. The maximum reevaluated flood elevation is 767.8 ft mean sea level (MSL) on the south side of safety-related structures, compared to the licensing basis flood-protection elevation of 773.7 ft MSL at that location.
Note a – See Attachment 5 for descriptions of the screening criteria.			

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source
Initial Preliminary Screening	C1. Event damage potential is less than events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009
	PS3. Design basis event mean frequency is less than 1E-5/y and the mean conditional core damage probability is less than 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009
	PS4. Bounding mean CDF is less than 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

The Duane Arnold FPIE/IF (Full-Power Internal Events and Internal Flood) and FPRA (fire PRA) models and associated documentation were reviewed for plant-specific modeling assumptions and related sources of uncertainty. The applicable lists of EPRI-identified generic sources of uncertainty in EPRI 1016737 (Reference 1) and EPRI 1026511 (Reference 2) were also reviewed.

Each PRA model includes an evaluation of potential sources of uncertainty for the base model using an approach consistent with the ASME/ANS RA-Sa-2009 requirements for the identification and characterization of uncertainties and assumptions (Reference 3). This evaluation identifies those sources of uncertainty that are important to the PRA results and may affect PRA applications. The process meets the intent of Steps C-1 and E-1 of NUREG-1855 (Reference 4).

The results of the base PRA evaluations were reviewed to determine which potential uncertainties could affect the 10 CFR 50.69 categorization process results. This evaluation meets the intent of the screening portion of Steps C-2 and E-2 of NUREG-1855, Revision 1.

Additionally, an evaluation of Level 2 internal events PRA model uncertainty was performed based on the guidance in NUREG-1855 (Reference 4) and EPRI Report 1026511 (Reference 2). The potential sources of model uncertainty in the Duane Arnold Level 2 PRA model were evaluated for the 32 Level 2 PRA topics outlined in EPRI 1026511.

The PDA 10 CFR 50.69 application also proposes use of the Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components described in EPRI 3002033536, Revision 1-A (Reference 8). The methodology incorporates the NRC-approved methodology originally submitted as EPRI 3002025288. The NRC approved the methodology for use as part of a 10 CFR 50.69 categorization process in the Safety Evaluation dated July 15, 2025 (Reference 9).

Prerequisite 1 of the Enhanced Passive methodology requires use of a technically adequate internal events PRA, including internal flooding, and requires review of key assumptions and sources of model uncertainty in the context of the pressure-boundary categorization application. Accordingly, the PRA assumptions and sources of uncertainty evaluated for this application were also reviewed for their potential to affect application of the Enhanced Passive methodology. The Enhanced Passive-specific uncertainty considerations are summarized below and are documented in greater detail in Reference 5.

The application-specific evaluation and disposition of these uncertainties, including supporting sensitivity studies, are documented in the PDA PRA LAR 50.69 Uncertainty Report (Reference 5).

For the 10 CFR 50.69 Program, the guidance in NEI 00-04 (Reference 6) specifies sensitivity studies to be conducted for each PRA model to address key sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty, such as human error, common-cause failure, and maintenance unavailability, do not mask SSC importance. RG 1.174, Revision 3 (Reference 7), cites NUREG-1855, Revision 1, as related

guidance. Section B of RG 1.174, Revision 3, acknowledges revisions to NUREG-1855 that expanded the discussion of uncertainties.

The identified PRA model sources of uncertainty were evaluated relative to this 50.69 application to determine whether additional application-specific sensitivity evaluations were needed. The resulting evaluations and sensitivity-study results are summarized in the table below and documented in Reference 5.

> Note: As part of the required 50.69 PRA categorization sensitivity cases directed by NEI 00-04, human-error and common-cause basic-event probabilities in the internal events/internal flood and fire PRA models are increased to their 95th-percentile values and decreased to their 5th-percentile values. In addition, maintenance-unavailability terms are set to 0.0. For the fire PRA model, a sensitivity case is also required that allows no credit for manual suppression. These sensitivity cases are capable of causing a component and its associated functions to be categorized as HSS. Therefore, uncertainty associated with modeled HEPs, CCF probabilities, maintenance unavailability, and manual fire suppression is addressed through the 50.69 application sensitivity evaluations.

The table below describes the PRA sources of model uncertainty, their potential impact on the 10 CFR 50.69 categorization results, and their application-specific dispositions.

PRA Source of Assumption/Uncertainty	Potential 10 CFR 50.69 Impact	Evaluation and 10 CFR 50.69 Disposition
Core Cooling Success Following Containment Failure or Venting The PDA PRA model does not credit HPCI, RCIC, LPCI, or Core Spray following containment failure, but credits selected alternate reactor vessel injection paths where appropriate. Uncertainty exists in the modelled likelihood of the applicable containment-failure challenge.	Changes in the modelled likelihood of containment failure could affect CDF, LERF, and the importance measures of SSCs credited in the affected accident sequences. This could potentially affect the preliminary safety-significance classification of applicable SSCs.	A sensitivity study was performed as described in Section 8.1 of Reference 5. The probability of the drywell-head rupture event was increased by a factor of two, from 6.00E-02 to 1.20E-01, and the model was requantified. The sensitivity resulted in an increase in CDF of approximately 1 percent of the base CDF and no change in LERF. Therefore, this uncertainty does not materially affect the PDA 50.69 categorization results and is not retained as a key source of model uncertainty for the PDA 50.69 application.

PRA Source of Assumption/Uncertainty	Potential 10 CFR 50.69 Impact	Evaluation and 10 CFR 50.69 Disposition
Digital Feedwater Control Failure Probabilities The PDA PRA model includes a basic event representing failure of the digital feedwater flow controller that causes the feedwater regulating valves to close. The modeled failure probability is based on reliability information from a similar-vendor study. Uncertainty exists regarding the applicability of this failure probability to the PDA digital feedwater control system.	Changes in the modeled digital feedwater control failure probability could affect CDF, LERF, and the importance measures of SSCs credited in the affected accident sequences. This could potentially affect the preliminary safety-significance classification of applicable SSCs.	A sensitivity study was performed as described in Section 8.2 of Reference 5. The probability of the digital feedwater flow controller failure event was increased by a factor of 10, from 2.44E-06 to 2.44E-05, and the model was requantified. The sensitivity resulted in no change in CDF or LERF. Therefore, this uncertainty does not materially affect the PDA 50.69 categorization results and is not retained as a key source of model uncertainty for the PDA 50.69 application.
Assumed Cable Routing The PDA fire PRA uses assumed cable-routing information where complete routing information is unavailable, including for certain non-safety-related equipment. These assumptions determine whether specific components and systems are modelled as available or unavailable in individual fire areas. Uncertainty exists regarding the applicability of the assumed routing to the actual plant configuration.	Changes in assumed cable routing could affect fire CDF, fire LERF, and the importance measures of SSCs credited in the affected fire scenarios. This could potentially affect the preliminary safety-significance classification of applicable SSCs.	A bounding sensitivity study was performed as described in Section 8.3 of Reference 5. In the sensitivity, assumed failures associated with uncertain cable routing were removed and the model was requantified. The sensitivity resulted in no change in CDF or LERF. Therefore, this uncertainty does not materially affect the PDA 50.69 categorization results and is not retained as a key source of model uncertainty for the PDA 50.69 application.
Enhanced Passive		
Applicability of PRA model assumptions and sources of uncertainty to Enhanced Passive categorization	Prerequisite 1 requires the analyst to review key assumptions and sources of model uncertainty in the context of the Enhanced Passive application, including potential non-conservatisms or use of methods not commonly accepted for risk-informed applications. Criterion 11 uses plant-specific CDF, LERF, CCDP, and CLERP results from the internal events and internal flooding PRA.	The generic and plant-specific internal events and internal flooding PRA assumptions and sources of uncertainty have been evaluated in Reference 5. Uncertainties having the potential to affect the PDA 50.69 application have been evaluated through the application-specific dispositions and sensitivity studies documented therein. No additional key source of model uncertainty specific to use of the Enhanced Passive methodology was identified.

PRA Source of Assumption/Uncertainty	Potential 10 CFR 50.69 Impact	Evaluation and 10 CFR 50.69 Disposition
Uncertainty in passive pressure-boundary failure frequency	Criterion 11 considers both risk metrics (CDF and LERF) and conditional consequence metrics (CCDP and CLERP). The approved methodology also includes provisions to address potential changes in passive component reliability associated with alternative RISC-3 treatment, including a cumulative sensitivity evaluation of applicable passive-component failure frequencies.	Section 3.2.2.4 of the NRC SE (Reference 9) concludes that Criterion 11, in combination with Prerequisites 1 and 2, provides reasonable confidence that sufficient safety margins are maintained and that potential increases in CDF and LERF resulting from changes in treatment are small. The cumulative sensitivity evaluation required by the Enhanced Passive methodology will be performed during implementation, as applicable. Therefore, this uncertainty is addressed through implementation of the approved Enhanced Passive methodology and the application-specific uncertainty evaluation documented in Reference 5.
Internal flooding modeling and protective features	Pressure-boundary failures can cause internal flooding and indirect failures. The Enhanced Passive methodology therefore requires a technically adequate internal flooding PRA and contains Prerequisite 3 for protective measures such as barriers, drains, and flood detection features that may not otherwise be explicitly modelled.	Internal flooding model uncertainty is evaluated in Section 2 of Reference 5. In addition, protective features credited in implementation of the Enhanced Passive methodology will be treated in accordance with Prerequisite 3 and will not be assigned LSS unless an evaluation demonstrates that loss of the protective feature would not cause the applicable pressure-boundary SSCs to exceed the Enhanced Passive HSS criteria. The NRC found this treatment consistent with 10 CFR 50.69(c)(1)(ii).
PRA configuration control and future changes	Changes to the plant, PRA, operating practices, or experience could alter the risk information used in Enhanced Passive categorization. Prerequisite 4 requires the PRA and supplementary analyses to reflect the as-built/as-operated plant and requires feedback and periodic update processes.	The PDA PRA configuration control and the 10 CFR 50.69 feedback and adjustment process will maintain the applicability of the risk information used for Enhanced Passive categorization. New or revised key sources of uncertainty that could affect categorization will be evaluated through these processes. The NRC found Prerequisite 4 consistent with 10 CFR 50.69(c)(1)(ii) and 50.69(e).

References

1. EPRI Technical Report 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," December 2008

2. EPRI Technical Update 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," December 2012

3. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009

4. NRC NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1, March 2017 (ML17062A466)

5. Report No. PDA-PRA-LAR-5069-UNCERT, "Assessment of Key Assumptions and Sources of Uncertainty for the Duane Arnold Energy Center PRA," August 2026

6. NEI Topical Report NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, July 2005 (ML052910035)

7. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (ML17317A256)

8. EPRI 2025 Technical Report 3002033536, "Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components (Revision 1-A)," August 2025 (ML25241A331)

9. NRC Safety Evaluation for EPRI Technical Report 3002025288, "Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components," July 15, 2025 (ML25168A015)